

Certified By

that it lacked adequate basis, was overly broad, failed to raise a concrete issue capable of litigation and raised matters which were inappropriate for consideration in this proceeding. The Staff specifically noted that Mr. Eddleman had shown no nexus between his specified generic issues and the Harris facility as required by Gulf States Utilities Company (River Bend Station, Units 1 and 2, ALAB-444, 6 NRC 760, 772 (1977)). "Staff Response to Supplemental Statements of Contentions By Petitioners To Intervene," dated June 22, 1982 at 57, 58. The Licensing Board deferred ruling on the admissibility of Contention 107 on the basis that it was premature and that Mr. Eddleman should proffer his contentions on generic issues when the Staff's SER was issued. LBP-82-119A, 16 NRC 2069, 2106 (1982). The Staff's SER was issued in November 1983. Appendix C of the SER set forth the Staff's consideration of generic safety issues which relate to the Harris facility. On January 17, 1984 Mr. Eddleman filed proffered contention 107-X, 107-Y, 107-2 on generic issues A-40, A-3, A-1, A-17, A-43, A-44, A-45, A-47 and A-49 as well as contentions on other aspects of the SER. On the same date by separate document, Mr. Eddleman filed a motion to defer his filing of contentions on generic safety issues. This response addresses that motion to defer and also moves the Board to deny Mr. Eddleman's contentions on generic safety issues. Our discussion follows. We respond to his separate contentions, 107X, 107Y and 107Z, related to generic safety issues in a separate filing. See "NRC Staff Response to Wells Eddleman's New Contentions Concerning the Staff's Safety Evaluation Report," dated February 16, 1984.

III. DISCUSSION

Since Mr. Eddleman filed Contention 107 in May 1982 and the Staff's response in June 1982, the Appeal Board has again had the matter of generic safety issues before it in Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2) ALAB-728, 17 NRC 777 (1983). In ALAB-728 at pages 806 and 807 the Appeal Board addressed the same sort of argument that Mr. Eddleman presents here, i.e., the Staff's analysis of generic safety issues is not adequate. The Appeal Board reviewed the history of its treatment of generic safety issues. The Appeal Board (17 NRC at 806) stated that to assist a Licensing Board to determine whether any serious safety issue exists the staff should include a discussion in the SER or a supplement thereto of unresolved safety issues that apply to the facility under consideration. At 17 NRC 807 the Appeal Board states:

"This being the case, the 'obligation' we placed on the staff to aid the adjudicatory boards runs to the boards and is not an obligation that is enforceable by a party to the operating license proceeding"
"[footnote omitted, emphasis added]"

* * *

An intervenor in an operating license proceeding is free to challenge directly an unresolved generic safety issue by filing a proper contention, but it may not proceed on the basis of allegations that the staff has somehow failed in its performance.

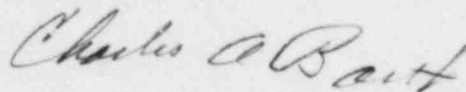
An examination of Mr. Eddleman's Contention 107 which is attached as Exhibit 1 conclusively shows that he is attempting to do exactly what the Appeal Board prohibits in Diablo Canyon cited supra. Mr. Eddleman is trying to raise the adequacy of the Staff's analysis of generic safety issues based upon the Appeal Board discussion in its view of the River Bend decision, ALAB-444, cited in his contention and referenced in

Appendix C of the Staff's SER. Pursuant to the Appeal Board's direction in Diablo Canyon, supra, Contention 107 should be denied and the derivative generic safety issue contentions separately filed by Mr. Eddleman on January 17, 1984 should also be denied. The Appeal Board's position in regard to generic safety issues is more fully set forth in our filing of this date upon Mr. Eddleman's contentions relating to the SER.

IV. CONCLUSION

The Appeal Board's decision in Diablo Canyon ALAB-728 prohibits the admission of Contention 107 which takes issue with the adequacy of the Staff's analysis of generic safety issues in Appendix C to the SER. The Licensing Board should not further defer Contention 107. The contention should be denied.

Sincerely,

A handwritten signature in cursive script, reading "Charles A. Barth".

Charles A. Barth
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 16th day of February, 1984

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#107 The SFR for SHNPP is deficient in that it does not provide valid assurance that SHNPP as built will be able to operate safely and within applicable NPC rules, IEEE and ASME codes, and other applicable requirements for operation, ~~including~~ without undue risk to the public health and safety resulting from any of the following under 12 NRC 683 (Byron) at 685-686, and while when the following problems which apply to SHNPP have not been adequately resolved:

(A) Water Hammers; (B) Westinghouse Steam Generator Tube Integrity (lack of it)(defective design SG's at Harris, like McGuire's and Summer's, included here); (C) Reactor Vessel Materials Toughness (assuring same for Harris' 1971-ASME-Code pressure vessels, coolant pumps, etc); (D) Fracture toughness of Steam Generator and Reactor Coolant Pump Supports (irradiation and frequent startups/shutdowns caused by defective steam generators or other management deficiencies of CP&L including failure to make appropriate modifications & repairs to assure safe & reliable operation, as at Brunswick); (E) Seismic design criteria (including collapse of cooling towers and piping thereto, shaking loose embrittled wiring and power insulation, and common-mode failures e.g. in the CPDM wiring which NPC does not have available to it on-site or otherwise met, & which I have not seen or had available); (F) escape of radiation via Containment Emergency Sump Performance (e.g. radionuclides released to containment from excessive numbers of reactor trips at Harris induced by steam generator design defects, or other CP&L design and operating deficiencies as described above); (G) Station Blackout from wiring insulation degradation or other causes; (H) failure to meet Shutdown Decay Heat-Removal Requirements due to loss of access to cooling towers, cooling tower failure

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on sabotage or terrorism, which Harris' security plan fails to provide adequate detection of and defense against)(or sabotage or terrorism against the RHR or other heat removal systems off SHNPP, which also Applicants' security plan does not provide adequate detection of, or defense against); (I)

Safety Implications of Control Systems (including failures cross-generational (of computer equipment) interfaces, due to the outdated design, untested software, and vulnerable wiring of Harris' Integrated Control System and other controls, failures in air lines due to, e.g. getting water in them during maintenance (CP&L's Brunswick maintenance is known to be sloppy) or tests, etc); (J) failure of hydrogen control measures, and effects of hydrogen burns on safety equipment (e.g. loss of power or control to Harris's only 2 hydrogen recombiners, failure of Harris's inadequate instrumentation to detect H_2 levels so the recombiners will be turned on before an explosion results, e.g. from operating motor sparking); (K) and Pressurized Thermal Shock (a serious problem at Robinson 2, see WPC documents thereon, nil-ductility temp already at 290 F after 11 years operating; applies to Harris due to outdated reactor pressure vessel, coolant pumps etc made to 1971 ASME-III codes) all of the above being as described by NRC, e.g. in NUREG-0606, Vol 1 No.1, Feb. 19, 1982 which document's descriptions of the above unresolved safety issues is incorporated herein by reference to ~~refix~~ show what the issues listed above are, through the "problem description" in NUREG-0606 for each issue listed above. (L) The concerns for interaction of such problems as expressed, e.g. by D.L. Basdekas of NRC (h-29-82 presentation to Commissioners and attachments thereto) also apply to Harris due to its old reactor vessel (ASME-1971 date) and control systems particular vulnerabilities. Interaction of such

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weaknesses in the vessel, primary system (as described and noted above, including contentions 47-51 incorporated herein by reference) and in the control system as described above, can compound accidents and lead to much more serious consequences to the public health and safety in the form of radiation releases, i.e. a minor accident or incident like loss of feedwater can escalate through a reactor vessel fracture and further control system failures to a completely uncontrolled release of radioactivity to the environment, e.g. by failure of containment isolation controls following vessel fracture or ATWS or any other incident (including ordinary reactor trips that lead to primary system relief valves releasing radioactivity inside containment in order to control primary system temperature and pressure). These matters include S/S Systems Interactions (Task A-17 of NUREG-0606, problem description incorporated by reference ~~xxxxxxxxxxxx~~ here for Task A-17.

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

Docket Nos. 50-400-0L
50-401-0L

John Runkle
Executive Coordinator
Conservation Counsel of North Carolina
307 Granville Rd.
Chapel Hill, NC 27514

Dr. Phyllis Lotchin
108 Bridle Run
Chapel Hill, NC 27514

Richard E. Jones, Esq.
Associate General Counsel
Carolina Power & Light Company
P.O. Box 1551
Raleigh, NC 27602

Atomic Safety and Licensing Appeal
Board Panel*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Docketing and Service Section*
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Bradley W. Jones, Esq.
Regional Counsel
USNRC, Region II
101 Marietta St., NW
Suite 2900
Atlanta, GA 30303

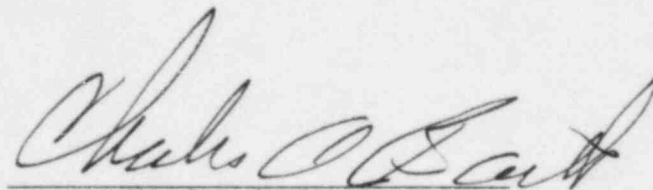
Ruthanne G. Miller, Esq.*
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, Dc 20555

Wells Eddleman
718-A Iredell Street
Durham, NC 27701

Robert P. Gruber
Executive Director
Public Staff - NCUC
P.O. Box 991
Raleigh, NC 27602

Atomic Safety and Licensing Board
Panel*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

George Trowbridge, Esq.
Thomas A. Baxter, Esq.
John H. O'Neill, Jr., Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, DC 20036



Charles A. Barth
Counsel for NRC Staff